Tennessee Valley Authority, Post Office Box 2000, Soddy-Daisy, Tennessee 37384-2000

May 14, 2007

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

10 CFR 50.73

Gentlemen:

TENNESSEE VALLEY AUTHORITY - SEQUOYAH NUCLEAR PLANT (SQN) UNIT 2 - DOCKET NO. 50-328 - FACILITY OPERATING LICENSE DPR-79 - LICENSEE EVENT REPORT (LER) 50-328/2007-002-00

The enclosed LER provides details concerning a manual reactor trip and automatic engineered safety feature (ESF) actuation of auxiliary feedwater. The manual trip occurred as a result of partial loss of main feedwater flow to the steam generators which resulted from a failure of a main feedwater pump control system. This report is being submitted in accordance with 10 CFR 50.73(a)(2)(iv) as an event that resulted in the automatic actuation of engineered safety features, including the reactor protection system.

Sincerely,

Glenn W. Morris

Glenn W. Morris
Manager, Site Licensing and
Industry Affairs

Enclosure
cc (Enclosure):
INPO Records Center
Institute of Nuclear Power Operations
700 Galleria Parkway, SE, Suite 100
Atlanta, Georgia 30339-5957

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NRC FORM 366 (6-2004) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104 EXPIRES: 06/30/2007
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

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4. TITLE
Manual Reactor Trip Following Partial Loss of Main Feedwater Flow

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
03	13	2007	2007	- 002 -	00	05	14	2007	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: *(Check all that apply)*			
1	☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
	☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
	☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
	☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
10. POWER LEVEL	☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
	☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
100	☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
	☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
	☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
N. R. Thomas, Nuclear Engineer	423-843-7749

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On March 13, 2007, at 1527 Eastern standard time with Unit 2 operating at 100 percent power, a manual reactor trip was initiated because of a partial loss of main feedwater flow to the steam generators. The immediate cause was failure of the 2A main feedwater pump control system. During normal power operations, Operations noticed that the steam generator levels on all four steam generators had begun to decrease. The 2A main feedwater pump speed controller output was noted to be decreasing. The controller was placed to manual but the controller failed to respond to the operator's attempt to increase output. The steam generator Loop 2 low-level alarm annunciated and the reactor was manually tripped. Following the reactor trip, the plant systems responded as designed. The unit entered Mode 3 and an event investigation was initiated. Troubleshooting determined the problem to be within the 2A main feedwater speed indicating controller. The root cause of the event was determined to be a faulty local/remote switch which is internal to the speed indicating controller.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

I. PLANT CONDITION(S)

Unit 2 was operating at 100 percent power when the reactor trip occurred.

II. DESCRIPTION OF EVENT

A. Event:

On March 13, 2007, at 1527 Eastern standard time (EST) with Unit 2 operating at 100 percent power, the reactor was manually tripped as a result of partial loss of main feedwater flow to the steam generators (EIIS code AB). The immediate cause was failure of the 2A main feedwater pump control system (EIIS code SJ). Troubleshooting determined the problem to be within the 2A main feedwater speed indicating controller. The root cause of the event was determined to be a faulty local/remote switch which is internal to the speed indicating controller.

B. Inoperable Structures, Components, or Systems that Contributed to the Event:

None.

C. Dates and Approximate Times of Major Occurrences:

March 13, 2007 at 15:22 EST	Operations noticed that all four steam generator levels had begun to decrease and then steam generator level deviation alarms were received.
March 13, 2007 at ~15:23 EST	Operations noted the 2A main feedwater pump speed controller output was at zero output and the pump was at minimum speed.
March 13, 2007 at ~15:24 EST	Operations placed the 2A main feedwater pump speed controller in manual and attempted to raise pump speed. The main feedwater pump speed did not increase.
March 13, 2007 at ~15:26 EST	Steam generator Loop 2 low-level alarm was received.
March 13, 2007 at ~15:27 EST	Operations initiated a manual reactor trip.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

D. Other Systems or Secondary Functions Affected:

No other systems or secondary functions were affected by this event.

E. Method of Discovery:

During normal power operations, Operations noticed that all four steam generators levels had begun to decrease. This was followed by the receipt of steam generator level deviation alarms. It was noticed that 2A main feedwater pump speed controller was at zero output and the 2A main feedwater pump was at minimum speed.

F. Operator Actions:

Operations placed the 2A main feedwater pump speed controller in manual and attempted to raise pump speed. The main feedwater pump speed did not increase. After the Loop 2 steam generator low-level alarm was received, the Senior Reactor Operator directed that a manual reactor trip be initiated. Control Room personnel stabilized the unit in a safe condition and maintained the unit in hot standby, Mode 3.

G. Safety System Responses:

The plant responded to the reactor trip as designed.

III. CAUSE OF THE EVENT

A. Immediate Cause:

The immediate cause of the event was loss of process control to the 2A main feedwater pump resulting in a reduction in steam generator level and a subsequent manual reactor trip.

B. Root Cause:

The root cause of the event was a faulty local/remote switch which is internal to the 2A main feedwater pump speed indicating controller. This switch was found to be erratic and of poor quality.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

C. Contributing Factor:

The K1 relay, which is internal to the main feedwater pump speed indicating controller, was determined to be a contributing cause. The K1 relay was found to be faulty.

IV. ANALYSIS OF THE EVENT

The plant systems responded to the reactor trip as designed. The reactor coolant system (RCS) average temperature was near 578.2 degrees F prior to the loss of main feedwater. Following the reactor trip, the loss of nuclear heat generation resulted in a rapid decrease in RCS average temperature to 539 degrees F. As heat removal in the steam generators decreased as a result of increased steam pressure, the decrease in RCS temperature slowed. The introduction of cold auxiliary feedwater (AFW) resulted in a slower, but continued reduction in RCS temperature until AFW flow was reduced after the reactor trip. RCS temperature then started to increase. RCS temperature remained within Technical Specification limits and bounded by the Safety Analysis Report (SAR) analysis.

The plant responded as expected for the conditions of the trip. No Technical Specification limits were exceeded and the SAR analysis of this event remained bounding.

V. ASSESSMENT OF SAFETY CONSEQUENCES

Based on the above "Analysis of The Event," this event did not adversely affect the health and safety of plant personnel or the general public.

VI. CORRECTIVE ACTIONS

A. Immediate Corrective Actions:

Control Room personnel responded as prescribed by emergency procedures. They diagnosed the plant condition and took necessary action to stabilize the unit in a safe condition. The 2A main feedwater speed controller was replaced by another controller that was procured, refurbished, bench calibrated, and installed in the plant.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

B. Corrective Actions to Prevent Recurrence:

Corrective actions will include implementation of a design change to jumper the local/remote switch out of the circuit for the applicable controllers. Applicable preventative maintenance procedures are being reviewed to cycle the K1 relays at appropriate frequencies.

VII. ADDITIONAL INFORMATION

A. Failed Components:

The 2A main feedwater pump speed indicating controller failed as a result of a faulty local/remote switch and K1 relay which is internal to the 2A main feedwater pump speed indicating controller.

B. Previous LERs on Similar Events:

There have been no LERs on similar events in the last three years.

C. Additional Information:

None.

D. Safety System Functional Failure:

This event did not result in a safety system functional failure in accordance with 10 CFR 50.73(a)(2)(v).

E. Loss of Normal Heat Removal Consideration:

This condition did not result in a loss of normal heat removal.

VIII. COMMITMENTS

None.